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December 22, 2005

SECY-05-0233

FOR: The Commissioners

FROM: Luis A. Reyes  
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SUBJECT: PLAN FOR DEVELOPING STATE-OF-THE ART REACTOR CONSEQUENCE  
ANALYSES

### PURPOSE:

The purpose of this paper is to inform the Commission of the staff's plan incorporating the combined efforts of the Offices of Nuclear Regulatory Research (RES), Nuclear Reactor Regulation (NRR), and Nuclear Security and Incident Response (NSIR) to (1) evaluate and update, as appropriate, analytical methods and models for realistic evaluation of severe accident progression and offsite consequences; (2) develop state-of-the-art reactor consequence assessments of severe accidents and update such analyses as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," dated December 1982; (3) identify mitigative measures that have the potential to significantly reduce risk or offsite consequences; and (4) develop an integrated, faster than real-time, computer-based tool to assist decision-making in the event of a severe reactor accident. This paper does not create any new commitments.

### DISCUSSION:

The evaluation of accident phenomena and offsite consequences of severe reactor accidents has been the subject of considerable research by the U.S. Nuclear Regulatory Commission (NRC). Most recently, with Commission guidance and as part of plant security assessments, the staff has concentrated on applying the accumulated research to perform analyses of severe accident progression and consequences, which are considerably more detailed, integrated, and realistic than past analyses. The results of these recent studies have confirmed and quantified what was suspected but not well-quantified — namely, that some past studies of plant response and offsite consequences could be extremely conservative, to the point that predictions were not

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useful for characterizing results or guiding public policy. In some cases, the overly conservative results were driven by the combination of conservative assumptions or boundary conditions; in other cases, simple bounding analysis was used in the belief that if the result was adequate to meet an overall risk goal, bounding estimates of consequences could be tolerated.

The subsequent misuse or misinterpretation of such bounding estimates further suggests that communication of risk attributable to severe reactor accidents should be based on realistic estimates of the more likely outcomes.

The staff is planning to perform consequence analysis for scenarios of radiological release frequency greater than or equal to  $10^{-6}$  per reactor year. If there are important security related events which are not captured by the spectrum of scenarios adopted for safety analysis, the staff, with Commission approval, can analyze those scenarios as part of the classified version of the study report. The analyses of such security related events should focus on additional mitigation as well as unmitigated consequences.

The staff has developed the attached plan to create a body of knowledge regarding the likely outcomes of severe reactor accidents, based on the most current emergency preparedness (EP) and plant capabilities and to identify reasonable and efficacious means by which to further mitigate such events. Through the evaluation of best available modeling and uncertainties, the staff also anticipates identifying opportunities for further efficient improvement and validation of modeling.

The basic approach will be to utilize the integrated modeling of accident progression (reactor and containment thermal-hydraulic and fission product response), which is embodied in the MELCOR code, coupled with modeling of offsite consequences (MACCS code) in a consistent manner (e.g., accident timing), drawn from our recent security assessments, to estimate offsite consequences for important classes of events. Toward that end, the staff will select events with appropriate consideration of probability. The staff will also perform offsite consequence analyses on a site-specific basis (reflecting site-specific population distributions and EP), although general accident progression modeling will be based on plant groupings by reactor and containment design types. In implementing this approach, it will be important to reflect all of the system and procedural plant improvements that have been incorporated as part of the industry's response to the NRC's security initiatives. Some additional analyses may also be needed to capture plant design specificities that would bear on severe accident probabilities or plant response.

The staff expects that the results of the reanalysis of severe accident consequences would provide the foundation for communicating that aspect of nuclear safety to Federal, State and Local authorities, licensees; and the general public. This evaluation of severe accident consequences would also update and replace the site-specific quantification of offsite consequences found in NUREG/CR-2239 and NUREG/CR-2723, "Estimates of the Financial Consequences of Reactor Accidents," dated September 1982. Publicly issued documents must incorporate effective risk communication and will be peer reviewed to ensure that objective is met.

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The development and application of an integrated, realistic methodology for use in assessing the consequences of hypothetical severe accident scenarios at individual reactor sites would also benefit the NRC's response to any real future events. The NRC's Operations Center does not currently have the capability to evaluate developing reactor scenarios using faster than real-time accident progression analysis directly coupled with consequence analysis.

Consequently, the Operations Center currently evaluates offsite projected doses using a generic radiological release (based on NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," dated February 1995), which is then adjusted (in a simplified way) based on available knowledge of containment status and systems operation. The staff concludes that the same modeling techniques used in developing state-of-the-art reactor consequences (as described in the enclosure to this paper) can also be used to enhance the NRC's capability to respond to real events and assist in training NRC personnel in preparing for such events. As part of the enclosed plan, the staff proposes an activity to develop a faster than real-time version of the coupled MELCOR and MACCS codes, which the Operations Center could use in evaluating reactor events. This would afford the capability to project the timing and progression of key events (e.g., steam generator level and boiloff, core water level and uncover, fission product release) and the alteration of the progression as a result of systems recovery and intervention. Offsite consequence estimates (dose projections, health effects, land contamination and costs) would also be available to decision-makers to further guide emergency response.

The overall schedule for this work will span approximately 3 years; however, selected higher-priority work will be scheduled for completion within the first year. Estimates (and documentation) of consequences for selected high-population and other reactor sites will be targeted for December 2006. Analyses to support development of preliminary design criteria for additional mitigation of offsite releases (i.e., beyond readily available measures) will be completed by May 2006. Analyses to evaluate the benefits of those additional active mitigation measures for specific accident scenarios will be completed by October 2006.

#### RESOURCES:

The staff estimates that the resource requirements of the proposed plan will be \$7.45M and 12 FTE spread over 3 years, if the project is fully funded starting in FY 2006.

For RES, the resource requirements are \$3,050K in FY 2006, \$2,700K in FY 2007 and \$1,700K in FY 2008, as well as 3.0 FTE per year.

For NRR, the resource requirements are 0.5 FTE in FY 2006, 0.25 FTE in FY 2007 and 0.25 FTE in FY 2008. None of the NRR resources are currently budgeted.

NSIR intends to support the project in FY2006 within existing budgeted resources. The combined 2 FTE required in FY 2007 and FY 2008 are not currently budgeted.

In FY 2006 RES has \$229K and 2.4 FTE in the budget, and \$2,821K and 0.6 FTE unbudgeted.

Unbudgeted resource requirements for RES and NRR in FY 2006 will be addressed through reallocation of lower priority work outside of new reactor licensing or, if necessary during the mid-year resource review process.

If funding is not available from mid-year, the staff is considering the list below of lower priority RES activities that may be displaced, deferred, or canceled, in order to fund the proposed plan.

- safety margin related to steam generator tube integrity,
- specific activities of the Generic Safety Issues Program,
- development and assessment of thermal-hydraulic tools,
- development and assessment of reactor fuel tools,
- development and assessment of containment and severe accident tools,
- support for licensing of mixed oxide fuel facility,
- reactor oversight process support,
- human factors research and regulatory support,
- materials aging models for passive component risk, and
- advancements in structure and earthquake engineering.

Detailed impacts of the portions that will be displaced, deferred, or canceled will be provided by a memorandum to the Commissioners in accordance with the agency's implementing procedures on reporting resource reallocations to the Commissioners.

In FY 2007, RES has budgeted \$1,500K and 3.0 FTE. Additional FY 2007 and FY 2008 resource needs for RES, NRR, and NSIR will be addressed in the FY 2008 Planning, Budget, and Performance Management process.

At the December 12, 2005, Closed Commission Meeting on security-related research, the Commission inquired as to how the staff could use additional funding to facilitate the development of this project. If contract funding becomes available, such funding would most effectively be used to minimize or eliminate the impact on the projects identified above for displacement, deferral or cancellation, and to initiate testing of the most promising strategies for additional offsite radiological release mitigation (e.g., area sprays for aerosol scrubbing). The current proposed program has experimental validation of such beyond readily available measures as a future, unfunded activity.

RES will have overall responsibility for project management and coordination, technical direction and support, and review. NRR will support scenario selection and probabilistic quantification, and will provide site-specific information, as needed. NSIR will provide technical direction and support for EP modeling, and will provide plant-specific information regarding security enhancements related to plant system and procedural modifications. Senior management oversight will be provided through an agency-level steering committee.

COORDINATION:

The Office of the General Counsel reviewed this package and has no legal objection. The Chief Financial Officer reviewed this package and determined that it has no financial impact.

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Enclosure:  
Plan to Develop State-of-the-Art Consequence Analyses

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## PLAN TO DEVELOPING STATE-OF-THE-ART CONSEQUENCE ANALYSES

### Goals

Assess the realistic consequences of a spectrum of risk-significant radiological releases to support safety- and security-related decision-making and to update such analyses as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," dated November 1982, which Sandia National Laboratory developed for the U.S. Nuclear Regulatory Commission (NRC).

### Objectives

Using a methodology based on state-of-the-art analytical tools, determine "best estimates" of the radiological dose consequences (including early and latent fatalities and land contamination) for each U.S. operating reactor site and present those results using risk communication techniques to achieve informed public understanding of the following factors:

- the extent and value of defense-in-depth features of plant design and operation, including mitigative strategies that are employed to reduce risk
- the most significant influential assumptions

As a starting point, the methodology to be used will reflect currently existing analytical research tools, coalesced into an integrated and coherent methodology to predict realistic outcomes. These analytical tools will be reviewed to identify potential substantive and cost-effective improvements that can be implemented in a timely manner, and those improvements will be incorporated. The staff will also identify potentially cost-beneficial areas for experimental validation using a systematic (internal NRC) process to identify the key influential analysis variables and assumptions, evaluate the degree of uncertainty associated with each, and determine the degree of cost and difficulty associated with reducing those uncertainties. Toward that end, the staff will use a catalog of available quantitative uncertainty results and other assumptions as the basis for identifying candidates for validation. The staff will also develop a research plan to guide continued substantive improvement, where possible, to the technical defensibility of the analytical tools developed and used in this study.

In addition, the staff will develop an integrated faster than real-time, computer-based decision-making tool, which can be used to enhance NRC and Federal responses to events of national significance. Toward that end, the staff will obtain input from computer code modeling efforts conducted by the Illinois Emergency Management Agency (IEMA).

Enclosure

**Potential Regulatory Uses Include:**

1. Improved Regulatory Analyses
  - a. backfitting decisions
  - b. rulemaking
  - c. prioritization and resolution of generic safety issues
  - d. identification of safety issues
  - e. resource allocation through the Planning, Budgeting, and Performance Management (PBPM) process
2. New and advanced reactor licensing and siting reviews
3. Emergency preparedness (EP) to assess the effectiveness of emergency action levels (EALs) and resolution of timing issues
4. Assessment of the effectiveness of proposed security mitigation strategies
5. Better informed public dialogue (with Federal, State and local authorities; licensees, and the public) on
  - a. reactor safety issue resolution
  - b. security issues assessment
  - c. new reactor design and siting reviews
6. Improved insights into licensees' current EP evacuation and sheltering strategies
7. To inform NRC's recommendations to DHS for beyond-readily-available mitigation strategies

**Summary**

The NRC staff has developed this plan to generate realistic release and consequence analyses for all nuclear power plant sites in the United States. To support more realistic and risk-informed regulatory decision-making, we will truncate scenarios that are considered extremely unlikely, so as not to obscure the value of preventive and mitigative features for the more likely scenarios. Hence, we will conduct consequence analyses only for scenarios that have a radiological release frequency (due to containment failure or containment bypass) greater than or equal to  $10^{-6}$  per reactor year. In addition, consistent with the Commission's direction in SRM dated May 14, 2003, the latent health effects analyses will cover a range of dose models. Specifically, the selected range will encompass models with thresholds from 0 to 5 Rem. \* *-yellow?*

Studies such as the NRC's Individual Plant Examination (IPE), Individual Plant Examination of External Events (IPEEE), and Simplified Plant Analysis Risk (SPAR) programs, as well as the Risk Analysis and Management for Critical Asset Protection (RAMCAP) study<sup>1</sup> conducted by the Electric Power Research Institute (EPRI), yield generic risk insights. Consequently, in developing state-of-the-art release and consequence analyses for all nuclear power plant sites in the United States, we will use the generic risk insights from these and other studies to identify generic accident scenarios. We will then calculate generic source terms and consequences for those analyses (using existing analyses where appropriate). Alternatively, we may use fuel damage classes (such as those identified in RAMCAP) to risk-inform the full spectrum of potential fuel damage classes that we will consider. We will also need to screen those fuel damage classes to focus on risk-significant scenarios.

The new assessment of accident consequences will consider (1) enhancements in plant design, operation, inspection, maintenance, and accident management; (2) security-related enhancements made by plant owners or implemented in response to requirements that the NRC has issued since the terrorist attacks on September 11, 2001; and (3) improvements in calculation methods for accident progression and consequences analyses. In order to reflect the plant enhancements, including EP modifications, made in response to risk assessments and security requirements, this new consequence assessment will require a coordinated effort by the Offices of Nuclear Regulatory Research (RES), Nuclear Reactor Regulation (NRR), and Nuclear Security and Incident Response (NSIR), to gather supporting plant-specific information from licensees. In some areas, such as EP, the assessment will require site-specific information, for which NSIR will have the lead responsibility. (Some sites have updated their evacuation time estimates, while other sites may rely on earlier conservative estimates that may need to be updated.) In other assessment areas, primarily those related to scenario development and systems response, sufficient information will be needed to ensure the applicability of the scenarios for each plant. NRR and NSIR will have the primary lead for providing this information, some of which may be solicited from licensees. RES will have the role of identifying specific information needs to support the accident consequence assessment.

In conducting this assessment, the staff will integrate existing state-of-the-art analytical tools, which the NRC is currently using, into a coherent methodology that can be used to better understand realistic best-estimate radiological dose consequences (including early and latent fatalities and land contamination) of any severe accident sequence selected for study. Generic MELCOR calculations, based on major plant classes of boiling- and pressurized-water reactors (BWRs and PWRs), will be used to determine the time to fuel failure, as well as the magnitude and timing of environmental fission product releases. We will then use these results to perform site-specific consequence evaluations for each risk-significant accident sequence identified for consideration.

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<sup>1</sup>The EPRI RAMCAP study developed generic fuel damage scenario classes for boiling- and pressurized-water reactors (BWRs and PWRs, respectively). Those scenario classes encompass loss-of-coolant accidents (LOCAs) without reactor pressure vessel (RPV) injection, small LOCAs without RPV injection, short-term station blackout, long-term station blackout, loss of offsite power, loss of ultimate heat sink, and loss of spent fuel pool integrity. Another class is designated for any plant-unique cases. These classes are generally consistent with other studies that identify risk-significant accident scenarios, such as plant-specific IPEs, as summarized in NUREG-1560, "Individual Plant Examination Program: Perspective on Reactor Safety and Plant Performance," dated December 1997.

The following table summarizes the overall process for developing state-of-the-art consequence analysis. To augment that summary, the next section presents a background discussion that compares the results of Sandia's 1982 Siting Study with those of recent security analyses. The remainder of this paper presents a detailed plan with milestones and costs for conducting the study.

### Overall Process for Developing State-of-the-Art Consequence Analysis

Process Step	Essential Consideration	Related Risk Reduction
<b>Identify representative severe accident scenarios:</b> Selection of scenarios based on risk insights, informed by EPRI RAMCAP generic fuel damage scenario classes and identification of risk-significance derived from site-specific IPE studies (NUREG-1560).  Anticipate general scenarios consistent with selected RAMCAP general fuel damage scenario classes.	<b>Selection of risk-significant scenarios:</b> <ul style="list-style-type: none"><li>radiological release frequency of greater than or equal to <math>10^{-6}</math> per reactor year</li><li>must reflect current plant design, layout, operation, accident management, and security enhancements</li></ul> Scenarios would be prioritized according to risk-significance.	Identify measures to reduce the likelihood of interfacing systems LOCA if radiological release frequency is $\geq 10^{-6}$ /RY and is a significant risk contributor.  Consider the contribution of security assessment plant improvements to reduce the likelihood of core damage.
<b>Evaluate representative source terms:</b> Accident progression, as well as the magnitude and form of fission product release	Perform MELCOR calculations (use existing calculations where appropriate) that employ best-practice calculation techniques, realistic models, and phenomenology to estimate the following:  <u>Accident Timing</u> <ul style="list-style-type: none"><li>Time to uncover the core</li><li>Time to core damage</li><li>Time to vessel failure</li><li>Time to containment failure</li></ul> <u>Environmental Source Term</u> <ul style="list-style-type: none"><li>Large early (prior to completion of evacuation)</li><li>Large late (after completion of evacuation)</li><li>Moderate (magnitude reduced by fission product retention)</li><li>Small (magnitude significantly reduced)</li></ul>	Identify measures to reduce the event likelihood, as well as measures to delay core damage or containment failure (where not evaluated in previous studies). In particular, identify measures to reduce the probability of bypass scenarios (e.g., interfacing systems LOCA), if radiological release frequency is $\geq 10^{-6}$ /RY and is a significant risk contributor.  Identify additional measures to enhance retention and scrubbing of fission products and estimate consequence reduction.

Process Step	Essential Consideration	Related Risk Reduction
Estimate consequences	<p>Perform site-specific MACCS2 calculations (use existing calculations where appropriate) that employ realistic models and phenomenology, best-practice calculation techniques, and site-specific EP to estimate:</p> <ul style="list-style-type: none"><li>• Early fatalities (Phase I)</li><li>• Latent cancer fatalities (Phase II)</li><li>• Environmental impact (Phase III)</li></ul> <p>Determine latent health effects for a spectrum of assumed thresholds.</p>	Identify EP features to ameliorate consequences.

## Background

The most recent security assessments indicate much smaller potential offsite consequences from severe accidents (e.g., fuel melt) than those often portrayed in earlier probabilistic risk assessments (PRAs) and consequence studies (e.g., NUREG/CR-2239, commonly referred to as "Sandia's 1982 Siting Study"). The smaller predicted consequences in the security assessments (compared to Sandia's 1982 Siting Study) are primarily attributable to the following factors:

- more realistic accident progression and consequence modeling
- more realistic emergency preparedness assumptions
- differences in the spectrum of accidents considered

The commonly cited consequences from the 1982 Siting Study relate to a very severe scenario designated as "siting source term one," or SST1. That report presents predicted offsite consequences for a distribution of weather conditions, including both mean values and low-probability weather conditions. Advocacy groups (such as Riverkeeper) usually cite the 99.5<sup>th</sup> or 99.9<sup>th</sup> percentile values. The use of these values corresponds to orders of magnitude increases in calculated early fatalities, often attributable to the large predicted effect of rainfall occurring at the worst possible time and washing out radionuclides over population centers. The obvious argument against using such low-probability outcomes is that using the 99.9<sup>th</sup> percentile outcome for an event that has a probability of  $10^{-6}$  per reactor year effectively transforms it into a likelihood of  $10^{-9}$  per reactor year. We should also reexamine the technical rigor and appropriateness of the analysis of washing out the radionuclides and resultant population exposure.

The following table compares the 1982 Siting Study consequences with those predicted in the recent security assessment of the reference BWR plant.

#### BWR Results

	1982 Siting Study, SST1 scenario (mean values)	1982 Siting study, SST1 scenario (99.9 <sup>th</sup> percentile)	Security assessment — bypass type scenario (mean value)	Security assessment — SBO type scenario (mean value)
Early fatalities	92	~15,000	0	0
Latent cancer fatalities	2662	>30,000	2,000 – 14,000 (depending on threshold)	70 – 5,000 (depending on threshold)

The differences between the predicted results are significant because of the factors cited above. The BWR bypass type scenario does not exhibit a great deal of difference in the severity of the radiological release. However, the 1982 Siting Study used a generic treatment of EP, assuming evacuation delay times of 1, 3, and 5 hours, with 30%, 40% and 30% probabilities, respectively. By contrast, the site-specific EP evaluation for the reference BWR determined that an evacuation delay time of 45 minutes was appropriate for that site. Thus, even the fast bypass type scenario would be mitigated by EP for the reference BWR. Moreover, the more slowly developing SBO type scenario showed a substantial margin in time available for EP. Similarly, for the more probable SBO-type scenario, the magnitude and timing of the radiological release is much less severe for the reference BWR than the Siting Study SST1 source term. In addition, the release timing for the SBO scenario is notably longer than for the SST1 source term. (SST1 release is assumed to begin in 1.5 hours.)

For PWR plants, the 1982 Siting Study is similarly dominated by the SST1 source term. In the security assessment, there was no fast bypass type scenario (comparable to that for the BWR), which might also have been comparable to an SST1 type release. In the security assessment, there was only a slowly developing SBO type scenario, with releases occurring much later than for the BWR. Containment failure and offsite releases did not occur for days. The table below compares predicted offsite consequences for the Siting Study with those predicted in the recent security assessment of the reference PWR plant.

#### PWR Results

	1982 Siting Study, SST1, (mean value)	1982 Siting Study, SST1, (99.9th percentile)	Security assessment — SBO type scenario (mean value)
Early fatalities	45	~20,000	0
Latent cancer fatalities	1,200	>20,000	0 – 70 (depending on threshold)

## **Study Conduct**

The proposed study will make full use of and capitalize on previous studies, take advantage of progress made in security-related work, and incorporate insights gained from recent safety-related studies.

With regard to scenario selection, we will review existing PRAs, IPEs, SPAR, and RAMCAP studies to identify representative severe accident scenarios for each type of plant (i.e., PWRs and BWRs). The scenario classes may encompass LOCAs without RPV injection, small LOCAs without RPV injection, short-term SBO, long-term SBO, loss of offsite power, and loss of ultimate heat sink. For any scenario with a radiological release frequency of less than  $10^{-6}$  per reactor year, no representative source term or consequence calculations will be performed.

In conducting this study, we will also factor in insights gained from extensive NRC research programs on containment performance and severe accident phenomenology. For example, we will incorporate insights from Sandia's testing of steel and pre-stressed concrete containment models, effect of containment leakage versus catastrophic containment failure; effects of measures to reduce the likelihood of liner melt-through of BWR Mark I containment, reduced likelihood of failure by direct containment heating, and so forth.

In conducting this study, we will also assign a representative source term to each scenario type. These source terms will be based on realistic MELCOR calculations, which account for the different depletion mechanisms in the reactor coolant and containment systems. Where appropriate, we will use existing MELCOR calculations, supplemented with additional calculations. Similarly, where appropriate, MELCOR analyses will parametrically credit readily available mitigation measures, as well as additional (i.e., beyond readily available measures) active systems (e.g., external spray, aerosol scavenging, or foam) to mitigate offsite releases.

Finally, with regard to the consequence analysis, we will use the MACCS2 code to generate site-specific consequences that account for weather conditions, population distribution/density, and EP (sheltering, relocation, and evacuation). We will report mean values for current plant configurations and selected scenarios that credit additional mitigative systems. The study will identify the value of these mechanisms that should inform decisions regarding future research to optimize system effectiveness through analytical studies and experimental verification. Phase I of the study will focus on early fatalities, while Phase II will address latent cancer fatalities, and Phase III will address land contamination.

## **Analysis Methods**

### *Identify Representative Severe Accident Scenarios*

Scenarios will be selected, for general classes of reactors, based on their contribution to risk. Each selected scenario will be reviewed to ensure that it would still be considered risk-significant, given recent plant modifications, current estimates of failure probabilities (e.g., pipe break frequency, emergency diesel reliability), and appropriate credit for emergency procedures and accident management strategies (including post-9/11 security enhancements). Scenarios that are determined to have an overall radiological release frequency less than  $10^{-6}$ /yr will be screened from consideration. Scenarios with a frequency that exceeds  $10^{-6}$ /yr will be ranked according to their risk-significance to ensure that those with the greatest contribution to overall risk are evaluated first.

#### *Evaluate Representative Source Terms*

MELCOR will be used to perform integral severe accident (in-vessel and ex-vessel progression, containment response, and fission product release, transport, and retention) analyses to supplement existing analyses. MELCOR input decks exist for the plants that were evaluated for security work, but can be modified for this study at reasonable cost. A Mark II containment model will be developed as part of this study. The models will need to be assessed to ensure that severe accident management strategies are appropriately credited.

MELCOR calculations will reflect current best practice modeling techniques and assumptions that represent the best estimate of accident progression for a given scenario (with no intentional conservatism built into the analyses). Prior to the start of the plant analyses, any potential substantive improvements to the modeling, which can be implemented in a timely cost-effective manner, will be identified and implemented. Phenomenological and sequence uncertainties, if considered important to overall risk (i.e., different assumptions might result in different accident progressions), will be treated. Calculations will consider research insights that would affect overall accident progression timing and source term magnitude (e.g., experimental insights regarding fission product release from Phébus and VERCORS programs; containment failure pressure, size, and location insights from the containment performance testing program; etc.). MELCOR calculations will also be performed to confirm the effectiveness of identified readily available additional measures to prevent core damage or containment failure, as well as to demonstrate the value of proposed beyond readily available measures (discussed later in this paper).

#### *Estimate Consequences*

Site-specific consequence calculations will be performed, with emphasis on realism that avoids needless conservatism. Site-specific data will be obtained for population distribution, meteorology, and parameters for modeling emergency response. The population distributions are readily available from the 2000 census data, using an existing NRC code that reads the Census Bureau files. Each plant has an evacuation time estimate for its 10-mile emergency planning zone (EPZ), and ad hoc emergency response will be considered beyond the EPZ. Licensees are required to measure certain meteorological parameters, and those files will be requested. If site-specific meteorology data are not readily available or retrievable in a timely manner, data from the nearest National Oceanic and Atmospheric Administration (NOAA) meteorology station may be substituted. For those States that have obtained potassium iodide (KI) for their residents, ingestion of KI will be included in the calculations. Consequence calculations will also be performed to confirm the adequacy of identified EP enhancements.

Several metrics will be considered for inclusion in the study, and a phased approach will be used in reporting the consequence results:

- Phase I – Early fatalities
- Phase II – Latent cancer fatalities
- Phase III – Land contamination and economic consequences

A new editing option has been added to MACCS2 to display the amount of land contaminated above a given value. Various options for the threshold “contaminated” value will also be investigated to provide perspective. It is anticipated that Phase III analyses will require Commission guidance on policy and criteria for responses to severe events. Prior to beginning site analyses, the MACCS code will be reviewed to evaluate any substantive, cost-effective modeling improvements which can be implemented in a timely manner, and those improvements will be implemented.

### **Identification and Valuation of Beyond Readily Available Mitigative Measures**

Based on the consequence analyses described above, the staff will identify mitigative measures that have the potential to significantly reduce risk or conditional consequences. Slowly breaking scenarios provide opportunities for mitigation. In contrast, fast breaking scenarios may be more effectively mitigated by preventive measures. Measures that may limit the quantity of released aerosols that are transported offsite will be considered. (These are briefly discussed below.)

Preliminary investigation of active mitigation concepts to limit offsite releases following containment failure has been performed as a follow-on to the pilot plant security assessments. Early evaluations have identified a set of fundamental characteristics that any mitigation concept must have to be effective:

- The mitigation option must have the ability to capture airborne radioactive aerosols as, or just after, they are released from a nuclear plant building (the release may not always come from the containment) and then retain the captured material on site.
- The mitigation option must be deployable independent of any onsite power source.
- The mitigation option must be operable in a high-radiation environment, as the initial release of noble gases could generate dose rates as high as 100s of rad/hr within 50 meters of the release point.
- The mitigation option must be effective in the outside ambient environment over a wide range of environmental conditions (e.g., wind speed, temperature, humidity).

These characteristics need to be confirmed to develop an optimal engineered solution. Because of the required investment for experimental verification of system efficacy and engineered design, it is prudent to perform initial analytical studies to evaluate the potential reduction in offsite release offered by proposed systems. Three promising candidates for active mitigation systems that could be deployed separately or in combination are use of fog or water sprays, use of foam technologies, and use of enhanced agglomeration agents, as follows:

- *Mitigation through Use of Fog or Water Sprays*

Water sprays would have the potential to scrub aerosol particles and condensable vapors from the plume in the vicinity of the release point, and would suppress the thermal buoyancy of the plume, thereby making the release easier to contain. Preliminary analyses show that control of an aerosol and vapor plume associated with a severe accident could be accomplished by spraying water at a rate that could be supplied by a few fire trucks. Assuming that the optimal water droplet size distribution and aerial flux could be achieved, this mitigation approach is appealing because of its simplicity and dependence upon readily available resources (i.e., water, pumps, and personnel trained in fire suppression).

- *Mitigation through Use of Foam Technologies*

Foam technologies have the potential to mitigate radioactive source term effects in a manner similar to sprays, but retain fission products within an onsite structure. It may be possible to deeply flood large buildings (from which a source term is emerging) with foams that could both suppress fires and provide fission product scrubbing and retention.

- *Mitigation through Use of Enhanced Agglomeration Agents*

The size distribution of the radioactive aerosol particles emerging from a damaged reactor system will likely be in the range of 0.5 to 5 micrometer, based on experimental studies and aerosol mechanics analyses. This size range is not optimal for the most effective water spray scrubbing, and considerable improvement in spray effectiveness could be gained if the target aerosol could be manipulated to larger sizes (in the range of 10 to 20 micrometers). This might be accomplished using enhanced agglomeration agents with larger inert particles. Potential candidates include fog mists or dense man-made smokes.

The assessment of beyond readily available measures for this project will consist of two parts. First, MELCOR scoping calculations will be performed to assess potential benefits and preliminary design criteria for beyond readily available mitigation measures. These calculations will adapt existing MELCOR models to estimate the source term reduction offered by active mitigation systems for selected scenarios. In pursuit of these goals, some additional standalone analyses will be performed to provide the basis for spray and foam mitigation effects to be factored into the MELCOR and MACCS2 assessments. Results of these MELCOR calculations will form the basis for identifying measures that show promise for significant source term reduction and that should be examined experimentally. Descriptions and cost estimates for experiments that would demonstrate the efficacy of any proposed measure would be produced as a part of these preliminary calculations. Second, these same mitigation measures would later be incorporated into MELCOR (and then MACCS2) calculations for select scenarios that are being evaluated as part of the effort for developing state-of-the-art consequence analysis. This will demonstrate the potential consequence reduction for risk-significant scenarios identified early in this study for a few high-population sites.

#### **Development of Faster than Real-Time Computer-Based Decision-Making Tool**

The staff will develop an integrated faster than real-time, computer-based decision-making tool, which can be used to enhance NRC and Federal responses to events of national significance. Toward that end, the staff will obtain input from computer code modeling efforts conducted by IEMA.

As a separate, but coordinated and parallel activity, the staff proposes to bring a best-estimate approach, founded on integrated modeling, to incident response and management. As a result of developing state-of-the-art consequence analyses described in this paper, we will have developed a large collection of accident signatures and plant models for the U.S. fleet. As part of the task to develop real-time decision-making capability, this accident signature database would be part of a developed software package that also includes MELCOR models or sub-models to provide the capability to interactively assess variations in those sequences to represent departures from the pre-calculated accident progression signatures. The capability to analyze variations in accident progression signatures would also include the ability to model the effects of intervention or mitigation. As part of this same decision-making capability, the code would model offsite health effects and land contamination using real event weather and site characteristics (including EP and variations). Both the MACCS code and other offsite models (e.g., RASCAL) would be considered for this offsite consequence modeling. (These codes are already quite fast-running.) As a tool for emergency management, this code will be faster than real-time and would be developed for use by NRC personnel in the Headquarters Operations Center. Training would also be provided for NRC personnel.

Another option would be to produce a simplified plant model capable of running faster than real-time. Presently, it is not clear how much runtime acceleration is possible through nodalization simplifications, etc., that preserve the essential timing accuracy of the predictions. Since significantly faster-than-real-time performance is desired, significant simplification would be required. This option should be investigated further as it allows use of many other MELCOR features, such as mitigative actions of sprays, regained plant systems, and so forth.

In addition to a faster-than-real-time capability to predict realistic source term signatures, an ability to evaluate the implications of this developing source term on emergency actions is also needed. This can be assisted by integrating the source term information with an atmospheric transport analysis tool [such as MACCS, RASCAL, or possibly one of the transport codes developed by Lawrence Livermore National Laboratory (LLNL)]. Desired features of this tool will be to accept time-dependent source term information from MELCOR and then make use of local topological information and weather data to predict likely local dose rate and land contamination information. This would require a puff release model capable of changing atmospheric transport direction and tendencies owing to terrain and wind changes. Attractive features in MACCS include dose assessment and land contamination prediction, whereas RASCAL includes a puff model for atmospheric transport. By contrast, the LLNL code suite has considerable national recognition for state-of-the-art capabilities. The feasibility of employing such codes should be assessed.

### **Project Organization**

- The Office of Nuclear Regulatory Research (RES) will have overall responsibility for project management and technical direction. The RES staff will be involved in all aspects of the effort and will be responsible for issuing the report.
- The Office of Nuclear Reactor Regulation (NRR) will be consulted in event selection and probabilistic quantification, and will provide the meteorology files for each site, along with certain plant data that may be needed to construct realistic MELCOR and MACCS2 input decks. NRR will also provide plant-specific information needed to confirm the appropriateness of scenario and system modeling.
- Scenario selection will be identified by the staff (RES and NRR) and discussed with the Advisory Committee on Reactor Safeguards (ACRS) and, if needed, a group of experts, to ensure soundness of the process.
- Contractor expertise will be required. Contractor staff will perform the MELCOR analyses for the various plant types, as well as the bulk of the MACCS2 consequence analyses. RES will conduct selected in-house MACCS2 calculations.
- The Office of Nuclear Security and Incident Response (NSIR) will provide the emergency response parameters used in the consequence analyses. Different values will be required for each site. NSIR will also provide generic and plant-specific information on security enhancements that relate to plant response for selected scenarios.
- Senior management oversight of the project will be provided through an agency-level steering committee. The steering committee will be briefed on progress, key technical and programmatic issues in directing project completion and success, and results as they become available. Steering committee review will be conducted every 6 months.

### **Commission Interactions**

The staff will keep the Commission informed annually and provide briefings to the Commissioners' Technical Assistants every 6 months.

The staff will also provide, for Commission consideration, options on the extent to which land contamination and offsite economic consequences should be addressed in developing recommendations for mitigative measures, and how best to achieve that objective.

## Cost<sup>2</sup>

The staff estimates that the contractor level of effort will total \$7.45M spread over 3 years, with a completion date of March 2009, as follows:

Developing State-of-the-Art consequences		+	Real-time decision-making tool	
FY 2006	\$2,550K		FY 2006	\$500K
FY 2007	\$2,200K		FY 2007	\$500K
FY 2008	\$1,200K		FY 2008	\$500K

The consequence calculations for high-population sites will be completed by October 2006. The additional experimental verification of mitigative system efficacy, if approved, will involve efforts to develop performance criteria for beyond readily available mitigation measures.

Budget Breakdown of Developing State-of-the-Art Consequences	Staff Months
Develop MELCOR plant models	22
Develop consistent MELCOR standard practice assumptions	2
Perform MELCOR Calculations	80
Develop updated EP modeling assumptions	20
MACCS2 analyses	33
Develop performance criteria for beyond readily available mitigation measures	3
MELCOR calculations to demonstrate active mitigation value	3
Reporting	29
<b>Total Project Duration</b>	<b>192</b>

The level of effort assumed for MELCOR analyses assumes roughly four base scenarios per reactor class, additional parametric calculations to treat important phenomenological uncertainties, and calculations to demonstrate the impact of identified readily available mitigation measures.

Experimental work to demonstrate the efficacy of mitigative strategies would follow if deemed valuable based on the analytical work performed in this project. Schedule and cost estimates would be produced as part of the effort during FY 2006.

The staff also anticipates that the proposed plan will require a total of approximately 12 staff full-time equivalents (FTE) over the 3-year period of the program for project management and coordination, technical direction and support, and review. This estimate includes 9 FTE from RES, 1 FTE from NRR, and 2 FTE from NSIR.

## Schedule

Expected Start Date:

Upon Approval of the Plan

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<sup>2</sup>The cost estimate shown is for contractor support and does not include the NRC staff FTE.

Expected Completion Dates<sup>3</sup>:

Performance Criteria for Beyond Readily Available Measures	05/20
High-Population BWR Mark I Sites	12/20
High-Population PWR Westinghouse 4-Loop, Large Dry Sites	12/20
Calculations to Demonstrate Active Mitigation Value	10/20
Project Completion	3/2006

Figures 1 and 2 provide additional schedule details.

**Deliverables**

1. A "living" collection of accumulated knowledge of physical experiment results, analytical studies, and computer codes that provide the bases for determining the most influential variables and assumptions, and that informs the nature and extent of the associated uncertainties.
2. A research plan, which identifies recommendations for cost-beneficial experimental or other research that should be undertaken to improve the accuracy of the analytical tools, or reduce uncertainties in key parameters that drive the associated risk. The plan should also estimate the completion of approved research, and incorporation of results into applicable codes and analysis methods.
3. A publicly available appendix to NUREG for traditional reactor safety risk analysis.
4. Both public and non-public versions of a supplement to the NUREG for security-related scenarios.
5. A library of pre-calculated high-fidelity analyses, using the MELCOR code, which cover the range of scenarios possible for plant designs in use in the United States.

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<sup>3</sup>Completion dates assume a project start date no later than January 3, 2006.